UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 400
ARLINGTON, TEXAS 76011-4005

February 6, 2008

Stewart B. Minahan, Vice
President-Nuclear and CNO
Nebraska Public Power District
72676 648A Avenue
Brownville, NE 68321

SUBJECT: COOPER NUCLEAR STATION - NRC INTEGRATED INSPECTION REPORT 05000298/2007005

Dear Mr. Minahan:

On December 31, 2007, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Cooper Nuclear Station. The enclosed integrated inspection report documents the inspection findings which were discussed on January 22, 2008, with Mr. M. Colomb, General Manager of Plant Operations, and other members of your staff.

This inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

Based on the results of this inspection, four findings were evaluated under the risk significance determination process as having very low safety significance (Green). All four of these findings were determined to be violations of NRC requirements. However, because these violations were of very low safety significance and the issues were entered into your corrective action program, the NRC is treating these findings as noncited violations, consistent with Section VI.A.1 of the NRC's Enforcement Policy. These noncited violations are described in the subject inspection report. If you contest the violations or significance of the violations, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN.: Document Control Desk, Washington, DC 20555-0001, with copies to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region IV, 611 Ryan Plaza Drive, Suite 400, Arlington, Texas, 76011-4005; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Cooper Nuclear Station facility.

In accordance with 10 CFR 2.390 of the NRC's Rules of Practice, a copy of this letter, its enclosure, and your response will be made available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Should you have any questions concerning this inspection, we will be pleased to discuss them with you.

Sincerely,

/RA/

Michael C. Hay, Chief
Project Branch C
Division of Reactor Projects

Docket: 50-298
License: DPR-46

Enclosure:
NRC Inspection Report 05000298/2007005
w/attachment

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SUNSI Review Completed: _WCW_ ADAMS: ☐ Yes ☐ No Initials: _WCW_____
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R:_REACTORS\CNS\2007\CNS2007-005RP-nht.wpd ADAMS ML080370616

SRI:DRP/C	C:SPE:DRP/C	C:DRS/EB1	C:DRS/PSB
NHTaylor	WCWalker	RLBywater	MPShannon
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2/5/2008	2/1/2008	2/1/2008	2/5/2008
C:DRS/OB	C:DRS/EB2	C:DRP/C	RI/DRP/C
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U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Docket:	50-298
License:	DPR-46
Report:	05000298/2007005
Licensee:	Nebraska Public Power District
Facility:	Cooper Nuclear Station
Location:	P.O. Box 98 Brownville, Nebraska
Dates:	September 23, 2007 through December 31, 2007
Inspectors:	N. Taylor, Senior Resident Inspector M. Chambers, Resident Inspector S. Garchow, Operations Engineer P. Elkmann, Senior Emergency Preparedness Inspector
Approved By:	M. Hay, Branch C, Division of Reactor Projects

Enclosure

TABLE OF CONTENTS

SUMMARY OF FINDINGS

IR 05000298/2007005; 09/23/2007 - 12/31/2007; Cooper Nuclear Station: Flood Protection, Postmaintenance Testing, Identification and Resolution of Problems.

The report covered a 3-month period of inspection by resident inspectors and region-based inspectors. Four Green noncited violations were identified. The significance of most findings is indicated by their color (Green, White, Yellow, or Red) using Inspection Manual Chapter 0609, “Significance Determination Process.” Findings for which the significance determination process does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, “Reactor Oversight Process,” Revision 3, dated July 2000.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Initiating Events

Green. A self-revealing noncited violation of Technical Specification 5.4.1.a, “Written Procedures,” was identified involving an inadequate procedure for controlling work on energized circuits. Specifically, inadequate procedural guidance in Administrative Procedure 0.9, “Tagout,” allowed power to be restored to the control logic for residual heat removal injection Valve RHR-MOV-27A while personnel were performing maintenance on the valve. This condition created a personnel hazard and resulted in the inadvertent opening of injection Valve RHR-MOV-25A due to interlock logic with Valve RHR-MOV-27A being satisfied. This issue was entered into the licensee’s corrective action program as Condition Report CR-CNS-2007-06844.

The finding is more than minor because it affects the equipment performance attribute of the initiating events cornerstone, and affects the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Using the Manual Chapter 0609, Phase 1 Screening Worksheet, the issue screened as having very low safety significance because the performance deficiency did not result in a condition that could have resulted in exceeding the Technical Specification limit for any reactor coolant system leakage or could have likely affected other mitigating systems causing a total loss of safety function. The cause of this finding is related to the human performance crosscutting component of work control in that the licensee did not appropriately coordinate work activities by incorporating guidance to consider the impact of changes to the work scope on other maintenance that was in progress [H.3(b)] (Section 4OA2).

Cornerstone: Mitigating Systems

- Green. The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” involving the failure to verify the adequacy of input assumptions to a design basis calculation. Specifically, a design basis control room flooding analysis assumed operators could terminate a turbine equipment cooling system pipe leak in the control room within 30 minutes when it is not possible to do so. This issue was entered into the licensee's corrective action program as Condition Report CR-CNS-2007-07708.

  The finding is more than minor because it affects the design control attribute of the mitigating systems cornerstone, and affects the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using Manual Chapter 0609, Appendix M, "Significance Determination Process Using Qualitative Criteria," a bounding quantitative analysis was performed resulting in the determination that the finding was of very low safety significance (Section 1R06).

- Green. A self-revealing noncited violation of Technical Specification 5.4.1. a, "Written Procedures," was identified because the licensee failed to establish an adequate postmaintenance test procedure to verify component performance following maintenance. Specifically, the licensee's postmaintenance test instructions were inadequate to verify an essential shutoff function of the Diesel Generator 1 day tank float valve following replacement on August 28, 2007. This issue was entered into the licensee's corrective action program as Condition Report CR-CNS-2007-07594.

  The finding is more than minor because it affects the procedure quality attribute of the mitigating systems cornerstone, and affects the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Though the failure of the float valve did impact operability of the diesel generator it would not have prevented the diesel generator, from starting and loading in response to an accident. Using the Manual Chapter 0609, Phase 1 Screening Worksheet, the issue screened as having very low safety significance because it did not represent a loss of safety system function. The cause of this finding is related to the human performance crosscutting component of resources in that the licensee's postmaintenance test procedure was inadequate to verify the essential shutoff function of the float valve [H.2(c)] (Section 1R19).

- Green. A self-revealing noncited violation of Technical Specification 5.4.1. a, "Written Procedures," was identified for the failure of maintenance personnel to follow procedures. Specifically, maintenance personnel failed to follow site administrative procedures that require verification of component identification prior to starting work. This resulted in maintenance personnel inadvertently attempting to remove a relief valve associated with the reactor equipment cooling system instead of the fuel pool cooling system. This error was identified while maintenance personnel were removing the wrong relief valve and an unexpected leak occurred. This issue was entered into the licensee's corrective action program as Condition Report CR-CNS-2007-07519.

  The finding is more than minor because it affects the configuration control attribute of the mitigating systems cornerstone and affects the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using the Manual Chapter 0609, Phase 1 Screening Worksheet, the issue screened as having very low safety significance because the maintenance personnel immediately restored the system integrity on noting the system leakage so that this did not represent a loss of safety system function. The cause of this finding is related to the human performance crosscutting component of work practices because maintenance personnel failed to implement an expected human error prevention technique [H.4(a)] (Section 40A2).

REPORT DETAILS

Summary of Plant Status

The plant began the inspection period at 100 percent power. On November 2, 2007, reactor power was reduced to approximately 70 percent for the purpose of planned maintenance. On November 3, 2007, the plant returned to full power. On November 18, 2007, power was reduced to approximately 70 percent due to a steam leak on the Reactor Feed Pump B. The leak was isolated and full power operation resumed on November 19, 2007, and continued through the end of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather (71111.01)

a. Inspection Scope

The inspectors completed a review of the licensee's readiness for seasonal susceptibilities involving extreme low temperatures. The inspectors: (1) reviewed plant procedures, the Updated Final Safety Analysis Report (UFSAR), and Technical Specifications (TS) to ensure that operator actions defined in adverse weather procedures maintained the readiness of essential systems; (2) walked down portions of the systems listed below to ensure that adverse weather protection features (heat tracing, space heaters, weatherized enclosures, etc.) were sufficient to support operability, including the ability to perform safe shutdown functions; (3) evaluated operator staffing levels to ensure the licensee could maintain the readiness of essential systems required by plant procedures; and (4) reviewed the corrective action program (CAP) to determine if the licensee identified and corrected problems related to adverse weather conditions.

- Service Water
- 125V and 250V Batteries

Documents reviewed by the inspectors included:

- Work Order (WO) 4542157
- General Operating Procedure 2.1.14, “Seasonal Weather Preparations,” Revision 9

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04Q)

.1 Partial System Walkdown

a. Inspection Scope

The inspectors: (1) walked down portions of the two risk important systems listed below and reviewed plant procedures and documents to verify that critical portions of the selected systems were correctly aligned; and (2) compared deficiencies identified during the walkdown to the licensee's UFSAR and the licensee's CAP to ensure problems were being identified and corrected.

- October 10, 2007, Residual Heat Removal Train B during Train A maintenance
- December 18, 2007, Standby Gas Treatment Train A during Train B maintenance

Documents reviewed by the inspectors included:

- System Operating Procedure 2.2A.SGT.DIV1, “Standby Gas Treatment System Component Checklist (Div 1),” Revision 3
- System Procedure 2.2.73, “Standby Gas Treatment System,” Revision 45
- System Operating Procedure 2.2A. RHR.DIV2, “Residual Heat Removal System Component Checklist (Div 2),” Revision 2

The inspectors completed two samples.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05Q)

a. Inspection Scope

The inspectors walked down the six plant areas listed below to assess the material condition of active and passive fire protection features and their operational lineup and readiness. The inspectors: (1) verified that transient combustibles and hot work activities were controlled in accordance with plant procedures; (2) observed the condition of fire detection devices to verify they remained functional; (3) observed fire suppression systems to verify they remained functional and that access to manual actuators was unobstructed; (4) verified that fire extinguishers and hose stations were provided at their designated locations and that they were in a satisfactory condition; (5) verified that passive fire protection features (electrical raceway barriers, fire doors, fire dampers, steel fire proofing, penetration seals, and oil collection systems) were in a satisfactory material condition; (6) verified that adequate compensatory measures were established for degraded or inoperable fire protection features and that the compensatory measures were commensurate with the significance of the deficiency; and (7) reviewed the UFSAR to determine if the licensee identified and corrected fire protection problems.

- November 2, 2007, Fire Zone 4A, Reactor Building Elevator and Accessway Area

- November 2, 2007, Fire Zone 6, Refueling Floor
- December 13, 2007, Fire Zone 4B, Reactor Building Heating, Ventilation and Control Area
- December 13, 2007, Fire Zone 4D, Reactor Motor Generator Set Oil Pump Area
- December 18, 2007, Fire Zone 2B, RHR Heat Exchanger 1A
- December 18, 2007, Fire Zone 2D, RHR Heat Exchanger 1B

Documents reviewed by the inspectors included:

- Cooper Nuclear Station Fire Hazards Analysis
- Fire Protection Safety Evaluation Report, May 23, 1979
- Administrative Procedure 0.23, "CNS Fire Protection Plan," Revision 49

The inspectors completed six samples.

b. Findings

No findings of significance were identified.

1R06 Flood Protection (71111.06)

a. Inspection Scope

Semi-annual Internal Flooding

The inspectors reviewed the flood protection features credited for protecting the control room from internal flooding sources. The review included: (1) the UFSAR, the flooding analysis, and plant procedures to assess susceptibilities involving internal flooding; (2) the UFSAR and CAP to determine if the licensee identified and corrected flooding problems; (3) operator actions for coping with flooding to ensure they can reasonably achieve the desired outcomes; and (4) a walk down of the control room to verify the adequacy of: (a) equipment seals located below the flood line, (b) floor and wall penetration seals, (c) door seals, (d) common drain lines and sumps, (e) sump pumps, level alarms, and control circuits, and (f) temporary or removable flood barriers.

Documents reviewed by the inspectors included:

- Design Criteria Document 38, "Internal Flooding System," November 8, 2006

The inspectors completed one sample.

b. Findings

Introduction: The inspectors identified a noncited (NCV) violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," involving the failure to verify the adequacy of an input assumption to a design basis calculation. Specifically, a design basis control room flooding analysis assumed operators could terminate leakage from a turbine equipment cooling system pipe leak in the control room within 30 minutes. This assumption was invalid, as the pipe break in question could not be isolated and the leakage terminated within the time required.

Description: The licensee documented an analysis of the impact of the most limiting medium-energy line break in the control room in design-basis Calculation NEDC 91-069, Revision 6. The piping system of concern for the analysis is a length of approximately 42 feet of 4-inch turbine equipment cooling (TEC) piping that runs vertically through the control room. The analysis assumed that the TEC line break could be identified and isolated by operations with 30 minutes, and that the resulting water level in the control room would be 1.12 inches. Given that the lowest piece of essential equipment is located 2.16 inches off the floor, this calculation demonstrated that a break in the TEC line in the control room would not affect equipment operability or prevent safe shutdown of the facility.

When challenged by the inspectors to demonstrate the isolation boundary for the TEC piping, the licensee discovered that over 40 valves would have to be located and manipulated to isolate the control room piping. The licensee determined that this would not be possible within 30 minutes. The licensee subsequently determined that the only way to stop the TEC leak in a timely manner would be to scram the reactor, trip the turbine, depressurize the TEC system by securing the TEC pumps, and then wait for the TEC system to gravity drain into the control room. The licensee demonstrated that if these actions were taken within 12 minutes, the ensuing system drain down would result in a water level of approximately 1.9 inches in the control room, still slightly below the level of the lowest piece of essential equipment.

The inspectors reviewed operations procedures for dealing with leaks from the TEC system, including Emergency Procedure 5.1BREAK, "Pipe Break Outside Secondary Containment," Revision 6, and Abnormal Procedure 2.4TEC, "TEC Abnormal," Revision 17. Neither of these procedures contained guidance to direct the operators to scram the reactor and shut down the TEC system in response to a TEC leak in the control room. In addition, these procedures did not provide operations with the knowledge that failure to take these actions in a timely manner could result in inoperability of mitigating systems or cause an event due to flooding the control room instrumentation panels.

The licensee entered this issue into the CAP as CR-CNS-2007-07708 and is currently considering a number of modification options to eliminate this run of unisolable piping in the control room.

Analysis: The performance deficiency associated with this finding was the failure to verify the adequacy of input assumptions to design basis calculations for a postulated pipe break in the control room. The finding is more than minor because it affects the design control attribute of the mitigating systems cornerstone, and affects the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using the Manual Chapter 0609, Appendix M, "Significance Determination Process Using Qualitative Criteria," the following assumptions were used in determination of final significance:

- The licensee demonstrated that the initiating event likelihood for a passive failure of the 42-foot length of unisolable TEC piping in the control room was 2.71E-7 per year.

- The licensee demonstrated that the TEC piping, originally designed as seismic class IIS, has been evaluated as seismic Class I-restrained, and as such would not be breached in a design basis earthquake.

With these bounding assumptions, the inspectors determined that the conditional core damage frequency related to this performance deficiency was less that 1E-6. Based on this information, the issue screened as having very low safety significance.

Enforcement: 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” requires that the licensee shall verify the adequacy of design basis information. Contrary to this requirement, the licensee failed to verify the accuracy of an input assumption to the control room flooding calculation. Specifically, the design basis control room flooding analysis assumed operators could terminate a turbine equipment cooling system pipe leak within 30 minutes when it was not possible to do so. Because this finding is of very low safety significance and has been entered into the licensee’s CAP as CR-CNS-2007-07708, this violation is being treated as an NCV consistent with Section VI.A of the Enforcement Policy: NCV 2007005-01, “Inaccurate Assumptions in Control Room Flooding Calculation.”

1R11 Licensed Operator Requalification (71111.11)

.1 Annual Inspection

a. Inspection Scope

The inspector performed an in-office review of the annual operating examination test results for 2007. Since this was the first half of the biennial requalification cycle, the licensee was not required to administer a written examination. These results were assessed against the standards of NUREG 1021, “Operator Licensing Examination Standards for Power Reactors,” and Manual Chapter 0609, Appendix I, “Operator Requalification Human Performance Significance Determination Process.” This review included the test results for 7 crews composed of a total of 39 licensed operators, which included: shift-standing senior operators, staff senior operators, shift-standing reactor operators, and staff reactor operators. There was 1 crew failure and 3 individual failures on the simulator. In addition, 1 individual failed the job performance measure of the annual requalification examination. The failures were remediated following the examination.

Documents reviewed by the inspectors included:

- Drill Scenario SKL052-52-89 (Bet 15035), Revision 1
- CR-CNS-2007-07028

The inspector completed one sample.

b. Findings

No findings of significance were identified.

.2 Quarterly Inspection by Resident Inspectors

a. Inspection Scope

The inspectors observed testing and training of senior reactor operators and reactor operators to identify deficiencies and discrepancies in the training, to assess operator performance, and to assess the evaluator's critique. The training scenario involved an control rod drop followed by a fuel failure and steam leak.

Documents reviewed by the inspectors included:

- Lesson Plan SKL052-52-89 (BET 15035), Revision 1
- CR-CNS-2007-07038

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule (711111.12Q)

a. Inspection Scope

The inspectors reviewed the maintenance effectiveness performance issues listed below to: (1) verify the appropriate handling of structure, system, and component (SSC) performance or condition problems; (2) verify the appropriate handling of degraded SSC functional performance; (3) evaluate the role of work practices and common cause problems; and (4) evaluate the handling of SSC issues reviewed under the requirements of the maintenance rule, 10 CFR Part 50, Appendix B, and the TSs.

- Service Water Booster Pump (SWBP) A bearing oil level high on September 1, 2007

- Failure of Diesel Generator 1 fuel oil system on September 11, 2007

The inspectors completed two samples.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Evaluation (71111.13)

a. Inspection Scope

The inspectors reviewed the two maintenance activities listed below to verify: (1) performance of risk assessments when required by 10 CFR 50.65 (a)(4) and licensee procedures prior to changes in plant configuration for maintenance activities and plant operations; (2) the accuracy, adequacy, and completeness of the information considered in the risk assessment; (3) that the licensee recognized, and/or entered as applicable, the appropriate licensee-established risk category according to the risk assessment results and licensee procedures; and (4) the licensee identified and corrected problems related to maintenance risk assessments.

- October 23, 2007, PC-AOV-237 relay replacement
- October 24, 2007, standby liquid control system relief valve replacement

Documents reviewed by the inspectors included:

- WO 4489525
- WO 4498765
- MP 7.3.24.1

The inspectors completed two samples

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors: (1) reviewed operator shift logs, emergent work documentation, deferred modifications, and standing orders to determine if an operability evaluation was warranted for degraded components; (2) referred to the UFSAR and other design basis documents to review the technical adequacy of licensee operability evaluations; (3) evaluated compensatory measures associated with operability evaluations; (4) determined degraded component impact on any TSs; (5) used the Significance Determination Process to evaluate the risk significance of degraded or inoperable equipment; and (6) verified that the licensee has identified and implemented appropriate corrective actions associated with degraded components.

The following equipment performance issue was reviewed:

- November 2, 2007, Drop in reactor equipment cooling surge tank level

Documents reviewed by the inspectors included:

- CR-CNS-2007-07624
- CR-CNS-2005-05588
- CR-CNS-2005-05556

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R19 Postmaintenance Testing (71111.19)

a. Inspection Scope

The inspectors selected six postmaintenance tests associated with the maintenance activities listed below for risk significant systems or components. For each item, the inspectors: (1) reviewed the applicable licensing basis and/or design basis documents to determine the safety functions; (2) evaluated the safety functions that may have been affected by the maintenance activity; and (3) reviewed the test procedure to ensure it adequately tested the safety function that may have been affected. The inspectors either witnessed or reviewed test data to verify that acceptance criteria were met, plant impacts were evaluated, test equipment was calibrated, procedures were followed, jumpers were properly controlled, the test data results were complete and accurate, the test equipment was removed, the system was properly re-aligned, and deficiencies during testing were documented. The inspectors also reviewed the UFSAR to determine if the licensee identified and corrected problems related to postmaintenance testing.

- August 30, 2007, Diesel Generator day Tank 1 float valve test following float valve replacement

- October 17, 2007, RHR power operated valve operability test

- October 23, 2007, Stroke test on PC-AOV-237 following relay replacement

- October 23, 2007, Stroke test on PC-AOV-238 following relay replacement
- October 24, 2007, Standby liquid control relief valve replacement
- November 3, 2007, Surveillance test of Service Water Pump C following overhaul

The inspectors completed six samples.

b. Findings

Introduction: A self-revealing NCV violation of TS 5.4.1.a, "Written Procedures," was identified because the licensee failed to establish an adequate postmaintenance test procedure to verify component performance following maintenance. Specifically, the licensee's postmaintenance test instructions were inadequate to verify an essential shutoff function of the Diesel Generator 1 day tank float valve following replacement on August 28, 2007.

Description: The licensee's diesel generator fuel oil transfer system consists of two storage tanks with two fuel oil transfer systems, each capable of filling either DG's day tank. The two fuel oil transfer systems are normally cross connected and when one tank is filled both transfer system headers are pressurized. The tank that is not being filled is isolated from filling by float Valve FOV-FLTV10, and a backup solenoid Valve DGDO-SOV-SSV5028. TS Surveillance Requirement (SR) 3.8.1.6, "Verify the fuel oil transfer system operates to automatically transfer fuel oil from storage tanks to the day tanks," and SR 3.8.3.1, "Verify the fuel oil storage tanks contain a combined volume of $\geq$ 49,500 gal of fuel," provide verification that there is an adequate and available inventory of fuel oil in the storage tanks to support a singe diesel generator's operation for 7 days at maximum post-LOCA load demand. The DG day tank float valves have an essential function to shut and prevent overflow of the day tank, ensuring an adequate 7-day supply of fuel oil for the emergency diesel generator (EDG) during an accident.

During a DG overhaul the last week of August 2007, DG 1 float Valve FOV-FLTV10 was replaced with a spare float valve from the licensee's warehouse. On August 30, 2007, postmaintenance testing of the replaced day tank float Valve FOV-FLTV10 incorrectly documented that the shut off capability of the replaced valve was tested satisfactorily per WO 4585698 postmaintenance test instructions. The testing did pressurize the float valve to check its shut function. However, there was no criteria for hold time or allowable day tank level rise to assure the shutoff function of the float valve was satisfactorily demonstrated.

On September 11, 2007, while filling the DG 2 fuel oil day tank and with the fuel oil fill system cross-connected to the DG 1 day tank, operations personnel received annuciators that indicated a rising level in the Diesel Generator 1 day tank. The unexpected level rise of Diesel Generator 1 day tank was due to the failure of float Valve FOV-FLTV10, and its backup solenoid Valve DGDO-SOV-SSV5028 to stop the flow of fuel into DG 1 day tank. This resulted in the licensee being unable to meet the requirements of SR 3.8.1.6 and potentially being unable to meet the requirements of SR 3.8.3.1. Operations personnel subsequently declared DG 1 inoperable.

The licensee's investigation of the failure to stop the flow of fuel into DG 1 day tank was documented in CR-CNS-2007-07594.

Analysis: The performance deficiency associated with this finding involved the licensee's failure to establish adequate postmaintenance instructions to verify the essential shutoff function of DG 1 day tank float Valve FOV-FLTV10. The finding is more than minor because it affects the procedure quality attribute of the mitigating systems cornerstone, and affects the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Though the failure of the float valve did impact operability of the diesel generator it would not have prevented the diesel generator from starting and loading in response to an accident. Using the Manual Chapter 0609, Phase 1 Screening Worksheet, the issue screened as having very low safety significance because it did not represent a loss of safety system function.

The cause of this finding is related to the human performance crosscutting component of resources in that the licensee's postmaintenance test procedure was inadequate to verify the essential shutoff function of the DG day tank float valve [H.2(c)].

Enforcement: TS 5.4.1(a) requires that written procedures be established, implemented, and maintained covering the activities specified in Regulatory Guide 1.33, Revision 2, Appendix A, dated February 1978. Regulatory Guide 1.33, Appendix A, Section 9.a, requires the development of procedures for maintenance activities. Contrary to the above, on August 30, 2007, the licensee failed to establish and implement an adequate postmaintenance test procedure to test the essential functions of float Valve FOV-FLTV10 to close and stop the flow of fuel oil into the DG 1 day tank. Because the finding is of very low safety significance and has been entered into the license's CAP as CR-CNS-2007-07594, this violation if being treated as an NCV consistent with Section VI.A of the Enforcement Policy: NCV 05000298/2007005-02, "Inadequate PMT Results in Inoperable Emergency Diesel Generator."

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors reviewed the UFSAR, procedure requirements, and TSs to ensure that the three surveillance activities listed below demonstrated that the SSCs tested were capable of performing their intended safety functions. The inspectors either witnessed or reviewed test data to verify that the following significant surveillance test attributes were adequate: (1) preconditioning; (2) evaluation of testing impact on the plant; (3) acceptance criteria; (4) test equipment; (5) procedures; (6) jumper/lifted lead controls; (7) test data; (8) testing frequency and method demonstrated TS operability; (9) test equipment removal; (10) restoration of plant systems; (11) fulfillment of ASME Code requirements; (12) engineering evaluations, root causes, and bases for returning tested SSCs not meeting the test acceptance criteria were correct; (13) reference setting data; and (14) annunciators and alarms setpoints. The inspectors also verified that the licensee identified and implemented any needed corrective actions associated with the surveillance testing.

- October 19, 2007, Residual Heat Removal Pump B in-service test
- October 24, 2007, Standby Liquid Control pump in-service test
- November 7, 2007, Service Water Pump A

Documents reviewed by the inspectors included:

- Surveillance Procedure 6.2RHR.101, "RHR Test Mode Surveillance Operation (IST)," Revision 22
- WO 4564111

- Surveillance Procedure 6.SLC.101, “SLC Pump Operability Test,” Revision 13

- Surveillance Procedure 6.1SW.101, “Service Water Surveillance Operation (Div1)(IST),” Revision 26

The inspectors completed three samples.

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP4 Emergency Action Level and Emergency Plan Changes (71114.04)

a. Inspection Scope

The inspectors performed an in-office review of Revision 36 to Cooper Nuclear Station Emergency Plan Implementing Procedure 5.7.1, “Emergency Classification,” implemented August 30, 2007. This revision updated the bases for flammable and toxic gasses in Emergency Action Levels 5.1.2 and 5.2.2, and made minor editorial corrections.

This revision was compared to its previous revision, to the criteria of NUREG-0654, “Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants,” Revision 1, and to the standards in 10 CFR 50.47(b) to determine if the revision adequately implemented the requirements of 10 CFR 50.54(q). This review was not documented in a safety evaluation report and did not constitute approval of licensee changes, therefore, this revision is subject to future inspection.

The inspector completed one sample during the inspection.

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation (71114.06)

a. Inspection Scope

The inspectors observed an emergency preparedness drill conducted on November 14, 2007. The observations were made in the control room simulator and the emergency operations facility and concentrated on the training evolution to identify any weaknesses and deficiencies in classification, notification, and protective action recommendation. In addition, the inspectors compared the identified weaknesses and deficiencies against licensee identified findings to determine whether the licensee is properly identifying deficiencies. Documents reviewed by the inspectors included:

- Emergency Plan for Cooper Nuclear Station, Revision 52

- Emergency Plan Implementing Procedures for Cooper Nuclear Station

- Emergency Preparedness Drill Scenario for November 14, 2007

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification (71151)

a. Inspection Scope

Mitigating Systems

The inspectors sampled licensee submittals for the PI listed below for the period October 2006 through September 2007. The definitions and guidance of Nuclear Energy Institute 99-02, "Regulatory Assessment Indicator Guideline," Revision 5, were used to verify the licensee's basis for reporting each data element in order to verify the accuracy of PI data reported during the assessment period. The inspectors reviewed licensee event reports, monthly operating reports, and operating logs as part of the assessment.

- Mitigating Systems Performance Index

The inspector completed one sample in this cornerstone.

b. Findings

No findings of significance were identified.

4OA2 Problem Identification and Resolution (71152)

.1 Routine Review of Identification and Resolution of Problems

a. Inspection Scope

The inspectors performed a daily screening of items entered into the licensee's CAP. This assessment was accomplished by reviewing condition reports (CRs) and WOs and attending corrective action review and work control meetings. The inspectors: (1) verified that equipment, human performance, and program issues were being identified by the licensee at an appropriate threshold and that the issues were entered into the CAP; (2) verified that corrective actions were commensurate with the significance of the issue; and (3) identified conditions that might warrant additional follow-up through other baseline inspection procedures.

b. Findings

No findings of significance were identified.

.2 Selected Issue Follow-up Inspection

a. Inspection Scope

In addition to the routine review, the inspectors selected the issues listed below for a more in-depth review. The inspectors considered the following during the review of the licensee's actions: (1) complete and accurate identification of the problem in a timely manner; (2) evaluation and disposition of operability/reportability issues; (3) consideration of extent of condition, generic implications, common cause, and previous occurrences; (4) classification and prioritization of the resolution of the problem; (5) identification of root and contributing causes of the problem; (6) identification of corrective actions; and (7) completion of corrective actions in a timely manner.

Documents reviewed by inspectors included:

- CR-CNS-2007-07519
- CR-CNS-2007-06844

The inspectors completed two samples during this inspection.

b. Findings

.1 Procedure Noncompliance Causes Reactor Equipment Cooling System Leakage

Introduction: A self-revealing NCV violation of TS 5.4.1.a, "Written Procedures," was identified for the failure of maintenance personnel to follow procedures. Specifically, maintenance personnel failed to follow site administrative procedures that require verification of component identification prior to starting work. This resulted in maintenance personnel inadvertently attempting to remove a relief valve associated with the reactor equipment cooling system instead of the fuel pool cooling (FPC) system. This error was identified while maintenance personnel were removing the wrong relief valve and an unexpected leak occurred.

Description: On October 30, 2007, work was incorrectly started on reactor equipment cooling (REC) relief Valve REC-RV-11 during the performance of WO 4498536. This WO was intended for FPC relief Valve FPC-RV-11. The maintenance personnel thought they had identified the proper valve by viewing the work area from outside the door to the locked high radiation area containing these valves. On later entering the contaminated high radiation area and checking the component identification tag, they only looked at the side of the tag labeled Valve RV-11 and failed to note that the other side of the tag had the system designator REC. The inspectors also were informed that the workers did not have a copy of the work procedure with the equipment identifier at the work site having left it at the entrance to the contaminated high radiation area. There was no clearance order isolating Valve REC-RV-11, and when the maintenance personnel started to unscrew the threaded relief valve from the piping, a small leak started. The maintenance personnel stopped, rechecked their work procedure, rechecked the component tag, and discovered they were working on the wrong valve. This resulted in a small REC leak and additional radiation dose to correct this error. Subsequent actions were taken to restore Valve REC-RV-11 prior to the leak affecting the reactor equipment cooling system operability.

Licensee Procedures 0-HU-TOOLS; 0.40, "Work Controls" 0.31, "Equipment Status Control," and 4.0.4, "Conduct of Maintenance," all contain procedural requirements to ensure proper components are worked on and configuration control is maintained. Contrary to these requirements, the maintenance personnel did not properly verify their work was performed on the component specified in the WO. The licensee's investigation, documented in CR-CNS-2007-07519, stated, "The apparent cause of this event is determined to be a failure of the mechanics to effectively use their human performance tools. Specifically, self/peer checking, verbal communications and prejob briefings were not effective. Additionally, the high radiation dose rates in the area contributed to the mechanics haste while performing the task."

The inspectors reviewed the licensee's root cause evaluation and determined that the licensee had correctly identified the causes of the event.

Analysis: The performance deficiency associated with this finding was failure to follow administrative procedures. The finding is more than minor because it affects the configuration control attribute of the mitigating systems cornerstone and affects the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the performance deficiency resulted in maintenance personnel working on the wrong system and initiating a small amount of system leakage. Using the Manual Chapter 0609, Phase 1 Screening Worksheet, the issue screened as having very low safety significance because the maintenance personnel immediately restored the system integrity on noting the system leakage so that this did not represent a loss of safety system function.

The cause of this finding is related to the human performance crosscutting component of work practices because maintenance personnel failed to implement an expected human error prevention technique [H.4(a)].

Enforcement: TS 5.4.1.a requires that written procedures be established, implemented, and maintained covering the activities specified in Regulatory Guide 1.33, Revision 2, Appendix A, dated February 1978. Regulatory Guide 1.33, Appendix A, Section 9.a, requires that maintenance affecting the performance of safety-related equipment should be performed in accordance with written procedures. Contrary to the above, on October 30, 2007, maintenance personnel failed to properly verify that maintenance was performed on the component specified in the WO resulting in inadvertently working on the wrong component. Because the finding is of very low safety significance and has been entered into the licensee's CAP as CR-CNS-2007-07519, this violation is being treated as an NCV consistent with Section VI.A of the Enforcement Policy: NCV 05000298/2007005-003, "Procedure Noncompliance Causes Reactor Equipment Cooling System Leakage."

.2 Inadequate Tagout Procedure Results in Inadvertent Stroke of Motor-Operated Valve

Introduction: A self-revealing NCV violation of TS 5.4.1.a, was identified involving an inadequate procedure for work on energized circuits. Specifically, the inadequate procedural guidance in Administrative Procedure 0.9 allowed power to be restored to the control logic for Valve RHR-MOV-27A during maintenance, resulting in the inadvertent opening of Valve RHR-MOV-25A.

Description: On October 2, 2007, the licensee tagged out several motor operated valves in the RHR system for planned maintenance. Two of the MOVs tagged out, Valves RHR-MOV-25A and RHR-MOV-27A, the RHR Loop A inboard and outboard injection valves respectively, contain control logic interlocks with each other. More specifically, the control logic for each of the two valves is designed such that one valve can only be remotely opened if the other is fully closed.

Following maintenance activities associated with Valve RHR-MOV-25A the work control center proceeded to clear tags on the valve to return it to service failing to recognize that clearing these tags would affect Valve RHR-MOV-27A. Maintenance on Valve RHR-MOV-27A, which was isolated using a separate clearance order, had progressed to the point that technicians were manipulating the internals of the MOV logic circuit. In addition, the work control center operators failed to appreciate that clearing tags and re-energizing Valve RHR-MOV-25A would reintroduce electrical power into Valve RHR-MOV-27A logic circuit. As a result of clearing tags on Valve RHR-MOV-25A, power was restored to Valve RHR-MOV-27A motor starter coincident with manual manipulation of Valve RHR-MOV-27A logic internals, satisfying the "open" logic and resulting in Valve RHR-MOV-25A stroking to the full open position.

The licensee conducted an apparent cause evaluation to understand the reasons for the inadvertent valve stroke. This evaluation, documented in CR-CNS-2007-06844, concluded that the error was caused by inadequate procedural guidance in Administrative Procedure 0.9, "Tagout," that did not require maintenance personnel to verify that a proper clearance order had been provided to eliminate all possible sources of power from Valve RHR-MOV-27A motor operator. Additionally, the licensee determined that Procedure 0.9 did not contain guidance to warn maintenance personnel about sources of power that would remain energized during maintenance activities. The inspectors reviewed the licensee's evaluation and agreed with the licensee's conclusion that the cause of the event was inadequate procedural guidance in Administrative Procedure 0.9, "Tagout."

Analysis: The performance deficiency associated with this finding was the failure by the licensee to develop an adequate procedure for controlling tagouts. The finding is more than minor because it affects the equipment performance attribute of the initiating events cornerstone, and affects the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Using the Manual Chapter 0609, Phase 1 Screen Worksheet, the issue screened as having very low safety significance because the performance deficiency did not result in a condition that could have resulted in exceeding the Technical Specification limit for reactor coolant system leakage or could have likely affected other mitigating system causing a total loss of safety function.

The cause of this finding is related to the human performance crosscutting component of work control in that the licensee did not appropriately coordinate work activities by incorporating guidance to consider the impact of changes to the work scope on other maintenance that was in progress (H.3(b)].

Enforcement: TS 5.4.1.a requires that written procedures be established, implemented, and maintained covering the activities specified in Regulatory Guide 1.33, Revision 2, Appendix A, dated February 1978. Regulatory Guide 1.33, Appendix A, section 1.c, requires instructions for the conduct of the tagout process. Contrary to the above, on October 2, 2007, the licensee failed to establish adequate procedural guidance for verifying and controlling clearance orders in the conduct of maintenance associated with Valves RHR-MOV-25A and RHR-MOV-27A. This failure resulted in the inadvertent opening of injection Valve RHR-MOV-25A. Because this finding is of very low safety significance and has been entered into the licensee's CAP as CR-CNS-2007-06844, this violation is being treated as an NCV consistent with Section VI.A of the Enforcement Policy: NCV 2007005-04, "Inadequate Tagout Procedure Results in Inadvertent Stroke of Motor-Operated Valve."

.3 Semiannual Trend Review

a. Inspection Scope

The inspectors completed a semiannual trend review of repetitive or closely related issues that were documented in corrective action documents, corrective maintenance documents, and the control room logs to identify trends that might indicate the existence of more safety significant issues. The inspectors' review covered the 12-month period between September 2006 and September 2007. When warranted, some of the samples expanded beyond those dates to fully assess the issue. The inspectors reviewed the following issues:

- Shifts of the no-break power panel
- Alert and notification system (ANS) siren failures
- Drywall nitrogen makeup requirements
- Feedwater heater trips during downpowers

The inspectors compared their results with the results contained in the licensee's routine trend reports. Corrective actions associated with a sample of the issues identified in the licensee's trend report were reviewed for adequacy. Documents reviewed by the inspectors are listed in the attachment.

b. Assessment and Observations

The inspectors evaluated the licensee's CAP trending methodology and observed that the licensee had performed detailed reviews of developing issues. The inspectors determined that the licensee had generally addressed each of the areas reviewed. In addition to the observations already documented by the licensee, the inspectors noted the following:

.1 ANS Siren Failures

During daily plant status, the inspectors noted an increasing trend in the number of ANS siren failures. Specifically, the inspectors noted that in the first 11 months of calendar year 2007, a total of seven siren failures had been reported in CAP versus just four in the previous year. In response to this trend, the licensee initiated CR-CNS-2007-07533 on October 31, 2007. One corrective action was initiated in response to this CR but was closed based on expected implementation of a siren replacement project. The inspectors noted that the siren replacement project is not included in the licensee's nuclear project action plan for the next 3 years. Based on discussions with licensee personnel, the project is awaiting funding and may be implemented sooner than that, but a specific schedule has not been established. In addition, a severe winter storm on December 11, 2007, resulted in the loss of 11 of the 24 sirens, most of which failed due to loss of electrical power from local power outages. The inspectors noted that the siren replacement project referenced in the closed corrective action will install an independent direct-current power supply on each siren. The inspectors noted that no open corrective action exists in CAP that will resolve the adverse trend in siren failures.

.2 Drywell Nitrogen Makeup Requirements

During daily control room panel walkdowns, the inspectors noted that the Drywell nitrogen makeup system frequently repressurizes the Drywell with nitrogen. The inspectors questioned the licensee to determine if an adverse trend existed. Based on conversations with licensee personnel and review of available nitrogen makeup data from the past 3 years, the inspectors determined that the licensee is aware that the rate at which nitrogen makeup is occurring increased following the Refueling Outage RE23 in November 2006. In addition, the inspectors noted that the frequency at which nitrogen makeup occurs has not changed appreciably since the outage, and that the local leak rate testing conducted in Refueling Outage RE23 demonstrated the operability of the containment volume. As such, the inspectors determined that this trend does not represent an operability concern, but the data does suggest that an unidentified leak path exists from the containment volume. This potential trend was identified in CR-CNS-2007-01277 on February 22, 2007. This CR was closed to trend with no corrective actions assigned. Since that time, licensee personnel have been tracking nitrogen leakage and have made several attempts to locate the source of the containment outleakage. The inspectors noted that the existing trend is not identified in the CAP.

.3 Feedwater Heater Trips During Downpowers

During daily plant status activities, the inspectors noted an adverse trend in feedwater heater trips during plant downpowers. Specifically, the inspectors noted that in the past year, the Feedwater Heater B-2 had tripped off-line during three different downpower events. The inspectors noted that each feedwater trip is in itself a minor reactivity addition event, as the loss of feedwater heating adds positive reactivity and has the potential the upset the stability of the plant. The inspectors reviewed the operations response to each feedwater trip and determined that the control room staff had appropriately reacted to each of the feedwater trips, and that none of the trips reviewed had caused a significant increase in reactor power. The inspectors noted that the licensee had initiated CR-CNS-2007-04617 to document the trend on July 5, 2007. This CR documented the licensee's determination that the physical configuration of the drain lines from the Heaters A-2 and B-2 was the reason for the susceptibility of the heaters to trip during downpower events. The evaluation did not, however, explain why this phenomenon has been observed 12 times in the past 5 years on the Heater B-2, but only twice on the Heater A-2. The evaluation went on to discuss that the ongoing feedwater heater replacement project will correct this behavior when the feedwater Heaters A-2 and B-2 are replaced during the Refueling Outage RE25 in 2009. The inspectors noted, however, that the CR was closed and that the existing adverse trend of feedwater heater trips is not being tracked in CAP. In addition, the inspectors noted that the completion of the feedwater heater modification is not associated with any open corrective action in CAP.

4OA3 Event Follow-up (71153)

(Closed) NOV 50-298/2007007-01: "Failure to Promptly Identify and Correct a Defective Diesel Generator Voltage Regulator Circuit Board"

On August 17, 2007, a final significance determination for a White finding and a Notice of Violation was issued involving the failure to promptly identify and correct a degraded condition involving a defective voltage regulator circuit board used in EDG 2. Specifically, following installation of the defective EDG 2 voltage regulator circuit board, the licensee failed to determine the cause of two high voltage conditions which occurred on November 13, 2006, and failed to take corrective action to preclude repetition. As a result, an additional high voltage condition occurred resulting in a failure of EDG 2 on January 18, 2007. This violation of NRC requirements and the corrective actions are discussed in detail in NRC Supplemental Inspection Report 05000298/2007010. This violation is closed.

4OA6 Meetings, Including Exit

On December 3, 2007, the inspector presented the results of the licensed operator annual requalification examination to Mr. Dan Sealock, Training Manager. The inspector confirmed that proprietary information was not provided or examined during the inspection.

On December 31, 2007, the inspector conducted a telephonic exit to present the results of the in-office inspection of licensee changes to emergency action levels to Mr. B. Murphy, Supervisor, Emergency Planning, who acknowledged the findings. The inspector confirmed that proprietary information was not provided or examined during the inspection.

On January 22, 2007, the NRC resident inspectors presented the results of the inspection activities to Mr. M. Colomb and other members of his staff who acknowledged the findings. The inspectors confirmed that proprietary information was not disclosed in this inspection report.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

J. Bebb, Security Manager
J. Bednar, Staff Health Physicist, Radiation Protection
V. Bhardwaj, Engineering Support Manager
D. Buman, Systems Engineering Manager
T. Carson, Maintenance Manager
M. Colomb, Plant Operations General Manager
R. Dyer, Heat Exchanger Program Engineer
J. Dykstra, Electrical Engineering Program Supervisor
T. Erickson, System Engineering Supervisor
R. Estrada, Corrective Action Program Manager
P. Fleming, Nuclear Safety Assurance Director
K. Garner, Radiological Operations Supervisor, Radiation Protection
T. Hough, Maintenance Rule Coordinator
J. Kelsay, Emergency Planning Specialist
G. Kline, Engineering Director
D. Madsen, Licensing Specialist
M. McCormack, Electrical Systems/I&C System Engineering Supervisor
E. McCutchen, Regulatory Affairs Senior Licensing Engineer
M. Metzger, System Engineer
S. Minahan, Vice President - Nuclear & Chief Nuclear Officer
B. Murphy, Emergency Planning Manager
D. Oshlo, Radiation Protection Manager
S. Rezab, Emergency Planning Specialist
T. Rients, Emergency Planning Specialist
A. Sarver, Balance of Plant Engineering Supervisor
T. Stevens, Design Engineering Manager
K. Thomas, Mechanical Programs Supervisor
D. Willis, Operations Manager

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000298/2007-005-001	NCV	Inaccurate Assumptions in Control Room Flooding Calculation (Section 1R06)
05000298/2007-005-002	NCV	Inadequate PMT Results in Operable Emergency Diesel Generator (Section 1R19)
05000298/2007-005-003	NCV	Procedure Noncompliance Causes Reactor Equipment Cooling System Leakage (Section 4OA2)
05000298/2007-005-004	NCV	Inadequate Tagout Procedure Results in Inadvertent Stroke of Motor Operated Valve (Section 4OA2)

Closed

05000298/200707-01	NCV	Failure to Promptly Identify and Correct a Defective Diesel Generator Voltage Regulator Circuit Board (Section 4OA3)

LIST OF DOCUMENTS REVIEWED

Section 1R06: Flood Protection

Design Criteria Document 38, November 8, 2006
NEDC 91-0169, Revision 6
CR-CNS-2007-07708
CR-CNS-2007-08241
CR-CNS-2004-04816
Jelco Drawing 2853-26, Revision 6
General Electric Drawing 115D6063, Revision 2
Cooper Nuclear Station Internal Flooding Notebook, PSA-012, November 1, 2007
Emergency Procedure 5.1BREAK, "Pipe Break Outside Secondary Containment," Revision 6
Abnormal Procedure 2.4TEC, "TEC Abnormal," Revision 17

Section 1R12: Maintenance Rule

CR-CNS-2007-05991
CR-CNS-2005-02732
CR-CNS-2005-05739
NPPD Notification 10387347, Clean SWBP Mechanical Seal Catch Basins
NPPD Work Order 4396529
NPPD Work Order 4540005
NPPD Notification 10545005, MR Functional Failure Evaluation of SW-P-BPA
CR-CNS-2005-02732

Section 1R19: Postmaintenance Testing

WOs

4458761	4498765	4565109
4489525	4498766	4585698

CR-CNS-2007-5915
CR-CNS-2007-5916
CR-CNS-2007-5923
CR-CNS-2007-5929
CR-CNS-2007-7.05 Postmaintenance Testing

Surveillance Procedure 6.1.DG.405 completed August 30, 2007

Surveillance Procedure 6.1.DG.301 completed August 31, 2007

Surveillance Procedure 6.1.SW.101, Section 4

Surveillance Procedure 6.2RHR.201 Power Operated Valve Operability Test
MP 7.3.24.1

Section 4OA2: Problem Identification and Resolution

CR-CNS-2007-07533

CR-CNS-2007-08426

CR-CNS-2007-01277

Primary Containment System Health Report

System Operating Procedure 2.2.60, Primary Containment Ventilation and Nitrogen Inerting System," Revision 78

CR-CNS-2007-07914

CR-CNS-2007-04617

LIST OF ACRONYMS

ALARA	as low as reasonably achievable
ANS	alert and notification system
CAP	corrective action program
CFR	Code of Federal Regulations
CR	condition report
EDG	emergency diesel generator
FIN	finding
FPC	fuel pooling cooling
IST	inservice test
LOCA	loss of coolant accident
NCV	noncited violation
NEI	Nuclear Energy Institute
RHR	residual heat removal
SSC	structure, system, and component
TEC	turbine equipment cooling
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
WO	work order